NON-PUBLIC?: N
ACCESSION #: 8908150263
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Clinton Power Station PAGE: 1 Of 7

DOCKET NUMBER: 05000461

TITLE: Mechanical Failure of Rubber Expansion Joint Between the "A" Low Pressure Turbine and the Main Condenser Results in Loss of Condenser Vacuum and Manual Reactor Scram
EVENT DATE: 07/14/89 LER #: 89-029-00 REPORT DATE: 08/09/89

OPERATING MODE: 1 POWER LEVEL: 039

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: D.R. Morris, Director - Plant TELEPHONE: 217-935-8881
Operations, extension 3205

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SG COMPONENT: EXJ MANUFACTURER: X999
X SA BHD W900
REPORTABLE NPRDS: N
N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 14, 1989, with the plant at thirty-nine percent reactor power, a manual reactor scram was initiated in anticipation of an automatic scram that was imminent because main condenser vacuum was decreasing. (A loss of main condenser vacuum causes a turbine trip and results in an automatic reactor scram.) The reactor had been operating at 100 percent reactor power when condenser vacuum began decreasing. After the reactor scram, Groups 2, 3 and 20 automatic containment isolations occurred because of low reactor vessel water level. Group 1 containment isolation valves were manually closed in anticipation of the automatic isolation that would occur because of low condenser vacuum. The cause of this event is attributed to a mechanical failure of the rubber expansion joint located between the "A" low pressure turbine and the main condenser. The expansion joint failed because of age, overtorquing of the attachment nuts of the expansion joint

clamp assembly, and steam exposure that resulted from a detached protective cover. Corrective action for this event included replacing the rubber expansion joints between both the "A" and "B" low pressure turbines and the main condenser, torquing the attachment nuts of the clamp assembly to vendor recommended values, and reinforcing the welds of the protective cover.

END OF ABSTRACT



DESCRIPTION OF EVENT

On July 14, 1989, with the plant in Mode 1 (POWER OPERATION) at 100 percent reactor RCT! power, main condenser COND! SG! vacuum began decreasing. As a result, plant operators placed the reactor mode switch HS! in the shutdown position to initiate a manual reactor scram.

At approximately 0730 hours on July 14, 1989, operators noted that Offgas System (OG) WF! flow had increased from sixty-five standard cubic feet per minute (scfm) to seventy-five scfm indicating air inleakage into the main condenser. As a result of the increased OG flow, operators were sent to investigate the source of the air inleakage.

By 0755 hours, output from the main generator TG! TB! had decreased by approximately thirty megawatts, OG flow had increased to 190 scfm and was still increasing, and main condenser vacuum had dropped below twenty-six inches of mercury. As a result, operators reduced reactor power by reducing Reactor Recirculation System (RR) AD! flow.

At 0758 hours, operators further reduced reactor power to approximately forty percent of Rated Thermal Power (RTP) by inserting the control rods and by further reducing RR flow.

At 0800 hours with reactor power at approximately thirty-nine percent and with main condenser vacuum at approximately twenty-three inches of mercury and decreasing, in anticipation of an automatic reactor scram that was imminent because of the decreasing condenser vacuum, operators placed the reactor mode switch in the shutdown position and initiated a manual reactor scram. (A loss of main condenser vacuum causes a turbine trip and results in an automatic reactor scram.) Subsequent to the reactor scram, the RR pumps automatically transferred from fast speed to slow speed, the "A" turbine-driven TRB! reactor feedwater SJ! pump p! (TDRFP) was manually tripped because of increasing reactor water level, and operators verified that all control rods were fully inserted into the reactor core.

At approximately 0801 hours, operators started the motor-driven (MO) reactor feedwater pump (MDRFP) with control of the pump provided by the startup level controller FIK!, and then tripped the "B" TDRFP. The reactor scram signal was reset.

At approximately 0802 hours, the main turbine was manually tripped and the main generator GEN! automatically tripped.

At 0818 hours, the "B" condenser vacuum pump was started and condenser vacuum stabilized at approximately 15.5 inches of mercury.



At this point, turbine SB! bypass valves V! were controlling the reactor vessel pressure, the MDRFP was controlling reactor vessel water level, and the startup level controller was controlling the MDRFP.

At 0840 hours, operators attempted to start the "A" condenser vacuum pump to increase vacuum but it would not start. Electrical maintenance technicians were sent to investigate why the pump failed to start. The investigation determined that the "A" condenser vacuum pump would not start because of damage to a fuse FU! holder. The fuse holder was repaired by 0958 hours and the "A" condenser vacuum pump was started. As a result of this, condenser vacuum increased to approximately 17.7 inches of mercury.

At approximately 1031 hours, condenser vacuum began decreasing. In anticipation of an automatic isolation of the main steam lines and drains (Group 1 containment isolation) because of low condenser vacuum, operators manually initiated the Group 1 isolation.

At 1039 hours, the Reactor Core Isolation Cooling System (RCIC) BN! was manually started in the tank-to-tank recirculation mode to control reactor vessel pressure.

Reactor vessel pressure increased slowly to 1050 pounds per square inch gauge (psig) and as a result, at 1047 hours, the Reactor Protection System (RPS) JC! high pressure channel D tripped. Operators then manually opened Safety Relief Valve (SRV) RV! 1B21-F051B to reduce reactor vessel pressure.

At 1048 hours, as a result of opening SRV 1B21-F051B, reactor vessel water level swelled to Level 8, high water level, causing the MDRFP and RCIC to automatically trip.

At 1050 hours, with reactor vessel pressure at 800 psig, SRV 1B21-F051B was manually closed causing a reactor vessel pressure/level oscillation. As a result, reactor vessel water level dropped to Level 3, low water level, and the RPS automatically initiated. (At this time, the reactor was shutdown with all control rods inserted.)

At 1054 hours, the MDRFP was restarted to restore reactor vessel water level. At 1056 hours, the RPS trip signal was reset. Subsequently, RCIC was restarted in the tank-to-tank recirculation mode.

At 1115 hours, the "A" Residual Heat Removal (RHR) (BO! system was started in the suppression pool cooling mode to maintain suppression pool temperature within specified limits.

At 1133 hours, to aid in the investigation of the cause of the main condenser vacuum leak, the MDRFP was shut down to reduce noise in the condensate heater bays.



At 1143 hours, operators shut the reactor feedwater inlet shutoff valves and began injecting Reactor Core Isolation Cooling (RCIC) BN! into the reactor to control reactor vessel level and pressure.

At 1204 hours, reactor pressure reached 1010 psig and, in response to this, operators manually cycled SRV 1B21-F041A to reduce reactor vessel pressure to 800 psig. As a result of opening SRV 1B21-F041A, reactor vessel water leve
swelled to Level 8 causing RCIC to automatically trip.

At 1206 hours, operators initiated a manual reactor scram signal in anticipation of the reactor vessel water level decrease to Level 3. At 1207 hours reactor vessel water level decreased to Level 3.

At approximately 1210 hours, RCIC was restarted and used to control reactor vessel pressure and to restore reactor vessel water level.

By approximately 1232 hours, reactor vessel pressure and level had stabilized by using RCIC and the Reactor Water Cleanup System (RWCU) CE).

Following operator actions to stabilize the reactor, actions were taken to place the reactor into Mode 4 (COLD SHUTDOWN).

In addition to the Croup 1 containment isolation previously described, Groups 2, 3 and 20 automatic containment isolations occurred. These additional isolations occurred at approximately 0801 hours because of a Level 3, low reactor water level, trip.

During this event, condenser suction isolation valve 1CA001A was found incorrectly in the closed position. Following this discovery, a system lineup was performed and no other deficiencies were noted. An investigation was conducted to determine how 1CA001A came to be out of its required position. The investigation did not identify a specific reason for this valve being in the closed position. The mispositioned valve had no impact on this event.

Additionally, during this event, operators noted that control room position indication for valves 1B21-F303B, Reactor Feedwater Pump Main Steam Isolation Valve, and 1E51-F066, RCIC Testable Check Valve Disk, did not reflect the actual positions of these valves and that 1B21-F303B had a valve body to bonnet leak. Maintenance Work Request (MWR) D01712 was initiated to correct the indication problem associated with 1E51-F066, MWR D13916 was initiated to correct the position indication problem associated with 1B21-F303B, and MWR D01526, previously initiated for a leak, will be used to track correction of the leak associated with 1B21- F303B. These position indication problems and the valve leak had no impact on this event.



During this event, Operators noted that the main control room control switch for the RCIC Injection Valve, 1E51-F013, is incorrectly labeled. The label for the switch identifies 1E51-F013 as a throttle valve however, this valve can not be throttled. The label for this valve will be corrected by Field Alteration C-F038. This incorrect label had no impact on this event.

During this event, a manway gasket BHD! in the Auxiliary Steam System SA! "A" Electrode Boiler BLR! failed. This gasket was repaired under MWR D14179. The failure of this gasket had no impact on this event.

Following the event, the Radiation Protection and Chemistry departments reported no abnormal readings, indicating that no fuel damage occurred during this event.

No other automatic or manually initiated safety system responses were necessary to place the plant in a safe and stable condition. No other equipment or components were inoperable at the start of this event such that their inoperable condition contributed to this event.

CAUSE OF EVENT

The cause of this event is attributed to the mechanical failure of the rubber expansion joint EXJ! located between the "A" low pressure turbine and the main condenser. The investigation of the cause of this event identified a tear approximately sixty inches long in the rubber expansion joint. Illinois Power has concluded that the expansion joint failed because of age, overtorquing of attachment nuts in the expansion joint clamp assembly, and steam exposure.

The expansion joint is normally protected by a metallic cover plate. During a planned plant maintenance outage that occurred in the fourth quarter of 1987, an inspection of the condenser identified that approximately five to six feet of the cover plate had become detached. This portion of the cover plate was found on top of condenser tube bundles. The cover plate had been installed in the same area as the expansion joint failure found during LER 89-029-00. The cause of the cover plate failure was attributed to failure of the cover plate tab welds. The cover plate was replaced during the 1987 maintenance outage and the tab welds were inspected and reinforced where necessary. Currently the tab welds are routinely checked during condenser inspections. Additionally, during the plant's first refueling outage, which began in January 1989, a minimum of seventeen feet of expansion joint, including the area of the detached cover plate, under the low pressure turbine hoods A and B, was visually inspected and no degradation of the expansion joint was noted. During the period of time that the cover plate was detached (sometime between initial plant operation and the 1987 maintenance outage) the expansion joint was directly exposed to steam.



CORRECTIVE ACTION

The rubber expansion joints between both the "A" and "B" low pressure turbines and the main condenser were replaced under MWR D14184. The attachment nuts of the expansion joint clamp assembly were torqued as recommended by the expansion joint manufacturer with the concurrence of the condenser vendor. The cover plate tab welds were reinforced as necessary. Cover plates were repaired/replaced as necessary. The work associated with this MWR was completed on July 25, 1989.

ANALYSIS OF EVENT

This event is reportable under the provisions of 10CFR50.73(a)(2)(iv) due to the initiations of the Reactor Protection System JC! and due to the Groups 1, 2, 3 and 20 containment isolations.

Assessment of the safety consequences and implications of this event indicates that this event was not nuclear safety significant. Prompt and correct operator action was taken by initiating a manual scram of the reactor before an automatic scram occurred because of a decrease in condenser vacuum. Subsequent equipment responses to the transient that followed the scram occurred as designed, or were manually initiated by operator action prior to their automatic initiation.

This transient was compared to similar transients described in the Updated

Safety Analysis Report and the Transient Safety Analysis Design Report GEZ-7355, and was found to be within the design basis of the plant.

ADDITIONAL INFORMATION

The rubber expansion joint that failed was manufactured by La Favorite Industries and was supplied to Clinton Power Station (CPS) by Westinghouse Corporation in accordance with Westinghouse drawing number 664B075. The rubber expansion joint is a commercial grade item installed in non-safety related applications.

The replacement rubber expansion joints were manufactured by Maryland Rubber Corporation and were supplied to CPS by Crane and Cochrane Environmental Systems in accordance with Westinghouse drawing 664B075.

The thickness of the expansion joint that failed was approximately 0.375 inch and the thickness of the replacement expansion joints is approximately 0.625 inch. Both expansion joints meet the design requirements of Westinghouse drawing number 664B075 which allows a maximum limitation of 0.625 inch for the thickness.



The manway gasket that failed during this event was manufacturer part number 40B209721G1-P3 manufactured by Welmon Thermal.

LER 88-019-00 and 87-050-00 discuss reactor scrams associated with loss of condenser vacuum.

For further information regarding this event, contact D. R. Morris, Director - Plant Operations at (217) 935-8881, extension 3205.

ATTACHMENT 1 TO 8908150263 PAGE 1 OF 1

U - 601508
L45-89(08-09)-LP
2C.220
ILLINOIS POWER COMPANY

CLINTON POWER STATION, P.O. BOX 678, CLINTON, ILLINOIS 61727

August 9, 1989

10CFR50.73

Docket No. 50-461

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Clinton Power Station - Unit 1
Licensee Event Report No. 89-029-00

Dear Sir:

Please find enclosed Licensee Event Report No. 89-029-00: Mechanical Failure of Rubber Expansion Joint Between the A Low Pressure Turbine and the Main Condenser Results in Loss of Condenser Vacuum and Manual Reactor Scram. This report is being submitted in accordance with the requirements of 10CFR50.73.

Sincerely yours,

D. L. Holzscher
Acting Manager -
Licensing and Safety

RSF/krm

Enclosure

cc: NRC Resident Office
NRC Region III, Regional Administrator
INPO Records Center
Illinois Department of Nuclear Safety
NRC Clinton Licensing Project Manager

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